ANALYTICAL AND EXPERIMENTAL INVESTIGATIONS OF THE PASSIVE HEAT TRANSPORT IN HTRs UNDER SEVERE ACCIDENT CONDITIONS

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Abstract

Thermodynamic accident analyses have been performed with computer simulation models to investigate core heatup sequences, sensitivity analyses, power variations, anticipated transients without scram, and core displacement considerations for probabilistic safety analyses (PSA) of small gas-cooled high-temperature reactors (e.g. HTR-Module). In worst case considerations where not only a loss of the active heat removal system is assumed but also a loss of the vessel cooling system, the heat would be transported into the surrounding concrete structure. In such a case the concrete would act as a natural long-term intermediate heat storage dissipating the heat through the concrete surface.

Large scale and reactor safety experiments have been performed to investigate passive heat transport mechanisms - which can cooldown a HTR core during severe accident conditions - for validation basis of computer simulation codes used for accident analyses. In general, the comparisons of experimental and analytical results with computer calculations of the heat transport codes are in good agreement.

1. INTRODUCTION

High-temperature gas-cooled reactors (HTGRs) are being developed in Germany, the United States, Japan, and several other countries. Especially noteworthy is the construction of the High Temperature Test Reactor (HTTR) which is currently well underway in Japan. Recently HTGR programmes have mainly been focused on the development of modular reactor designs with some hundred MWt which mainly rely on passive systems to achieve a high degree of safety.

The results of probabilistic safety analyses (PSA) have shown, that the risk of an HTR-Module is smaller than that of a PWR with a comparatively scaled-up power /1/. Accident sequences with a probability for major release less than 10^{-8}/yr have been excluded due to supposed small risk contributions.

A safety goal of the HTR-Module (200 MWt) for beyond-design accidents (about $\leq 10^{-6}$/yr) is that radiation impacts are within the range of limits set by the Radiation Protection Ordinance for design basis accidents; evacuation of the public should not be regarded as necessary to avoid health effects. Furthermore, an extended safety philosophy is in discussion. Safety concepts for future reactor designs have presently been proposed which can exclude catastrophic consequences for the public by almost a deterministic approach taking into account severest accident scenarios; e.g. advanced safety enclosures (especially containment) for PWR and FBR /2/. Burst-proof cast iron pressure vessels, for example, are being considered for the HTR in context of the question: How safe is safe enough?

With this background, analyses of severest core cooling accidents with CORE HEATUP sequences and important temperature loadings of safety barriers have been performed for the HTR-Module design with regard to: (A) Sensitivities of accident analyses; (B) Accuracies of analytical methods; and (C) Importance for the safety concept.

2. SENSITIVITIES OF ACCIDENT ANALYSES

The dominant safety barrier of modular HTRs are the coated particles within the graphite fuel element. Important temperature loadings of the safety barriers are shown in Tab. 1 for core cooling accidents in events with total failure of active systems and passive heat removal dissipation by: natural convection, heat conduction, and thermal radiation. For the heat transport analyses analytical methods /3,4,5,6/ have been used in connection with experimental test facilities /7,8/ to investigate the reliability and sensitivity of computer simulation results (Tab. 2). Based on deterministic assumptions a severest accident scenario initiated by anticipated transients without scram (ATWS) is discussed in the following as worst case consideration:

(1) Failure of active feedwater supply for steam generator (SG) and for vessel cooling system (VCS);

(2) Failure of the blower trip and blower shut-off-valve;

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Tab 1 : Core Cooling Accidents with Main Temperature Loads of the Safety Barriers During Heat Removal

Barrier	Accident	Design Value
Fuel Element (FE)	• Loss - of - Coolant Accident (LOCA)	< 1600 °C
Pressure Vessel (PV)	• Loss - of - Forced Convection Transients (LOFT) • Anticipated Transients without Scram (ATWS)	< 350 °C
Concrete Cell (CC) Confinement (CON)	• Loss - of - vessel Cooling System (LOVS) • Primary System Ruptures (PSR)	< 100 °C

Tab 2 : Analytical and Experimental Investigations of the Core Cooling Accidents for the HTR - Module (200 MW)

Accident	Analysis	Experiment
ATWS	Influence of Scram Time THERMIX / KONVEK / KISMET / SIKADE	(Heat Transfer)
LOFT	Safety Margins THERMIX / KONVEK / KISMET	LUNA - 2D
LOCA	Safety Margins THERMIX / HEATING	HTA - 5 LUNA - 3D
LOVS PSR	Sensitivity of Core Arrangement THERMIX / RALOC	(Concrete Behavior)

(3) Failure of the control rod systems with delayed reactor trip;

(4) Power-, pressure-, and temperature transients can cause:

(a) Loss-of-coolant accident (LOCA); the reactor is depressurized without the VCS;

(b) Loss-of-forced convection transients (LOFT); the reactor is pressurized without VCS; and

(c) ATWS consequences; primary system ruptures cannot be excluded for the long-term behavior.

The accident scenario has been analysed separately for a HTR module unit for electricity generation; it is also typical for a process heat application /9/. The probability of accident sequences is expected to be less than 10^{-8} per reactor year.

2.1 Evaluation of Safety Margins

The consequences of ATWS depend decisively on the time at which the reactor is tripped and what decay heat level (DHL) is reached afterwards (power input). To examine the main influence on the reactor temperature response, two power transients have been investigated /10/ as sensitivity analyses with inherent reactor trip (case 2) by negative temperature coefficient (DHL after 1250 sec) and with delayed reactor trip (case 1) by control rods (DHL after 20 sec). The resulting pressure transients in the primary system are shown in Fig. 1.

The fast pressure and temperature transients would lead in ATWS case 2 to a temperature-induced failure of the large safety valve in open position and/or to a failure of the blower due to high temperatures in the primary circuit which increase during 5 min over 600 °C. The temperature behavior of the depressurized reactor in a LOCA is discussed later.

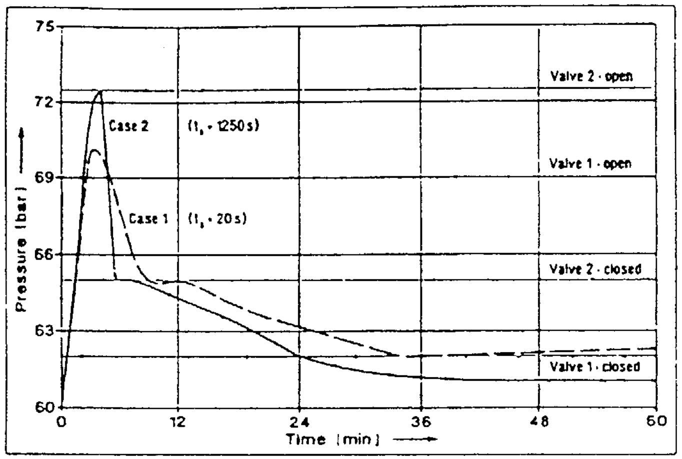

Fig. 1 : ATWS Pressure Transients in the Primary Circuit

The pressure peak could be reduced in ATWS case 1 by the small safety valve from 70 to 62 bar. The temperature transients in the primary circuit are shown in Fig. 2; after the initial temperature maximum they increase slowly during the first hours nearly to 500 °C due to the power input by the blower. A delayed blower failure can be avoided by countermeasures so that the reactor remains under pressurized conditions.

The long-term temperature behavior of the PRESSURIZED REACTOR during a LOFT and without operating vessel cooling system is shown in Fig. 3. Best estimate and conservative calculations of the core temperatures are far below the design value for the fuel element of 1250 °C and show great safety margins. A delayed failure of the pressure vessel which is heated-up during a week at about 500 °C can be avoided by countermeasures for which several days are available (Fig. 4).

Core maximum temperatures of sensitivity analyses for the long-term temperature behavior of the DEPRESSURIZED REACTOR under LOCA conditions are compiled in Tab. 3; they are not significantly influenced by the vessel cooling system. Best estimate core maximum peak temperature is 1440 °C after about 1 day and rises to

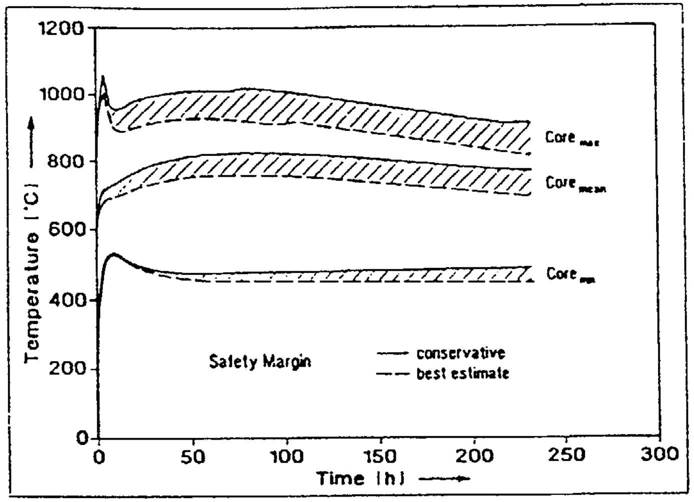

Fig. 3 · LOFT Temperature Transients in the Core

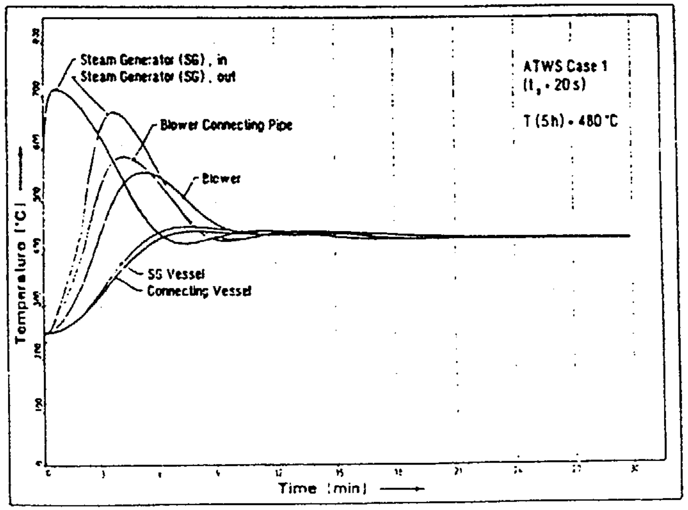

Fig. 2 · ATWS Temperature Transients in the Primary Components

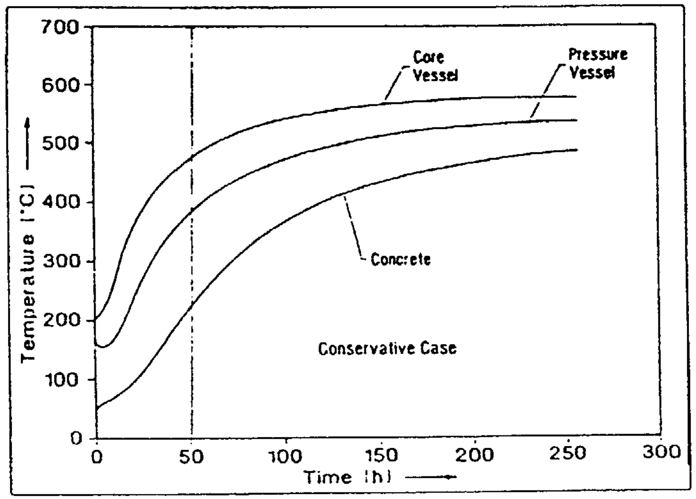

Fig. 4 · LOFT Max. Temperature Transients in the Pressure Vessel and Concrete Cell

Tab. 3 : LOCA Safety Margins and Sensitivities Related to the Max. Core Temperature

Fast Depressurization	Max. Core Temperature [°C]	Difference [°C]	[%]
Best Estimate	1440	-	--
Deviations : + 10% Decay Heat - 10% Eff. Conductivity - 10% Heat Capacity	 1510 1485 1455	 70 45 15	 5 3 1
Conservative (all Deviations together)	1570	130	9

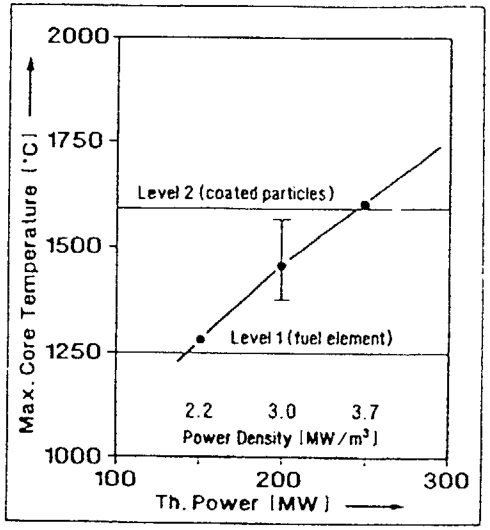

Fig. 5 : LOCA Influence of Thermal Power on the Max. Core Temperature

1570 °C with conservatisms for main influencing parameters which could be reduced on an updated data base. About 50 °C are to be added because of ATWS initial starting conditions with the result of 1620 °C in total. The temperature peak value represents a great safety margin for a significant fission product release out of the fuel elements within the core during LOCA conditions based on experimental heating tests /11/.

The sensitivity influence of POWER VARIATIONS for the 200 MWt module unit on core maximum peak temperature in a LOCA with vessel cooling is shown in Fig. 5 for best estimate conditions. A thermal power of 250 MW leads accordingly to a peak temperature of 1615 °C in some per cent of the fuel elements as a hot spot in the core; 10 % of the core is exposed to temperatures over 1400 °C and 30 % over 1250 °C during a period of some days (Fig. 6).

The peak temperature of 350 °C occurs in the pressure vessel after about one week and remains below failure limits (Fig. 7). The heat is removed in the radial direction mainly by thermal radiation - besides convection and conduction - from the pressure vessel surface to the vessel cooling system at the concrete of the primary cell (50 °C). The sensitivity influence of the thermal radiation by an emissivity variation of 0.6 to 0.8 reduces the vessel temperature respectively.

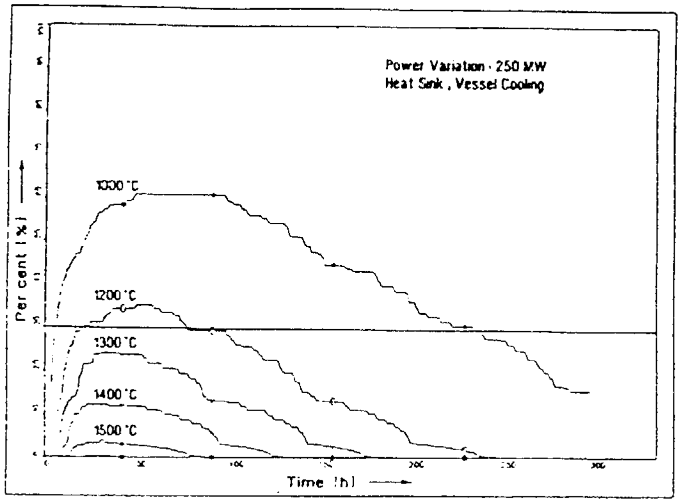

Fig. 6 : LOCA Temperature Distribution in the Fuel Elements

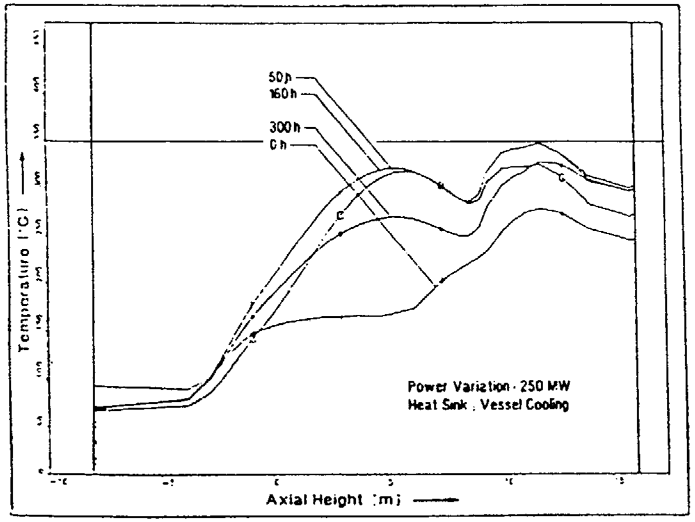

Fig. 7 : LOCA Temperature Profile in the Reactor Pressure Vessel

With failure of the active water-cooled VESSEL COOLING SYSTEM - as assumed in the ATWS scenario - the heat is transported during a LOCA into the surrounding concrete structure of the primary cell. The temperature response of the reactor components was analysed including the confinement as a time dependent boundary condition. In the thermal properties of the concrete, energy absorbing processes like release of hydrated water and phase changes were taken into account /12/. The resulting temperature transients (1 °C/hour) showed that the heated concrete structure (250 °C/meter) acts as a long-term natural intermediate heat storage dissipating the heat during some months through the concrete surface in the radial direction at core midheight.

2.2 Evaluation of Heat Dissipation Perturbations

Different scenarios with postulated ruptures of the primary system can be considered in an ATWS post accident phase. The following CORE DISPLACEMENT CASES have been analysed to examine the sensitivity of heat dissipation for great perturbations (Fig. 8) based on the assumptions: Case A rupture of the fuel discharge tube in the pressure vessel, case B with a core dissemination, and case C comparatively with a

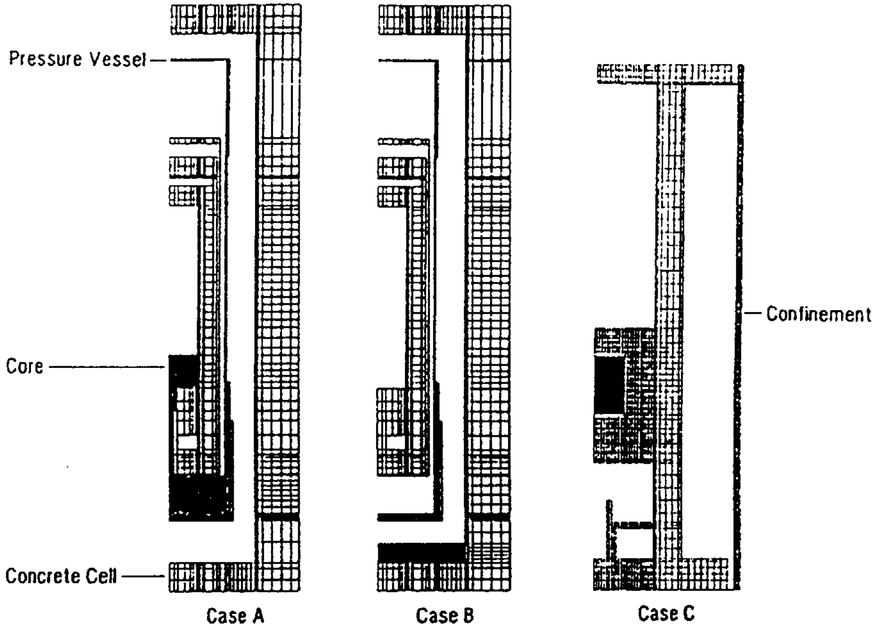

Fig. 8 : Core Displacement Scenario A, B and C in the Concrete Cell

compact core. The corresponding core maximum temperature transients of the scenario cases A (1560 °C), B (1125 °C), and C (1690 °C) are shown in Fig. 9 (regardless of pressure loadings and recriticality aspects).

The important aspect of case A is that the heat dissipation would partly be interrupted due to the insulation of the pressure vessel with a temperature-induced failure so that case A results in a typical core displacement of case B. For the concrete of the bottom foundation temperatures over 750 °C are calculated after several weeks in case B.

The propagation of premixed corrosion products - released hydrated water/steam and reactions with hot graphite surfaces - into the confinement atmosphere has not been investigated; conditions for hydrogen combustion modes cannot be excluded and are to be analysed. Suitable computer codes are being established. The temperature-induced consequences could be decisively reduced - as a precaution - by an additional carbon brick layer between the pressure vessel and the concrete foundation structure. The heat dissipation in the radial direction is more effective than in the axial direction (e.g. case C with lower concrete temperatures).

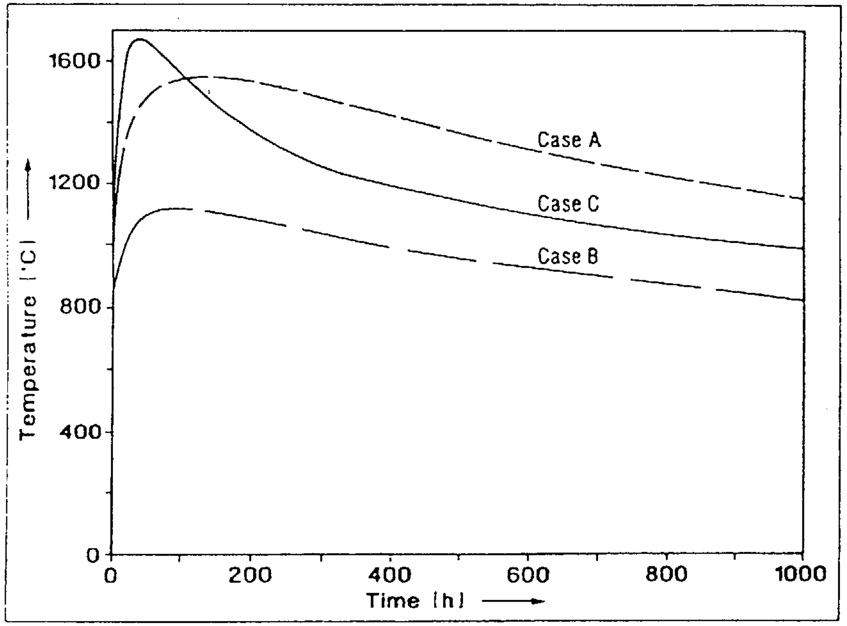

Fig. 9 : Core Maximum Temperatures for Scenario A, B, C

3. ACCURACIES OF ANALYTICAL METHODS

Pretest and posttest calculations have been performed to investigate passive heat transport mechanisms in HTRs to provide a validation basis of computer codes used in accident analyses: (a) Large scale experiments with the LUNA test facility to simulate the heat transport in the primary loop under LOFT conditions; and (b) Reactor safety experiments with the AVR reactor to simulate the heat transport in the reactor components under LOCA conditions.

3.1 Comparison with LOFT results

The LUNA loop is the only test facility which has been used to simulate heat transport by natural convection processes in HTR primary circuits in detail (Fig. 10). Observing geometrical and thermodynamic similarity conditions, different primary circuit arrangements (HTR-500, THTR-300, HTR-100, and HTR-Module) have been tested /13/. In

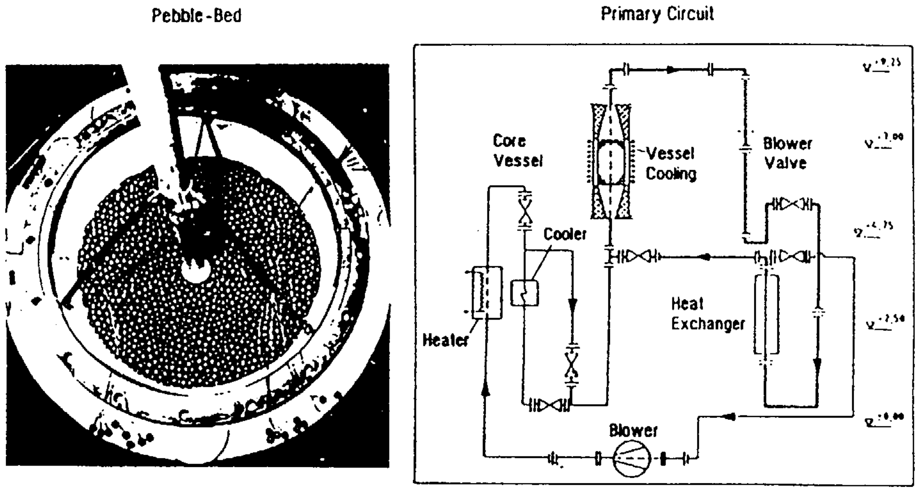

Fig. 10 : LUNA Experimental Facility for Loop Circulation of Natural Convection in HTRs (p = 40 bar, T = 300 °C)

general, the LUNA results can be well predicted by the THERMIX-C-2D code with some exceptions because of modelling (3-D effects) and numerical (small pressure drops) limitations. Fig. 11 shows a comparison of the pretest calculation with core internal natural convection and Fig. 12 the posttest calculation with natural loop circulation for the modular design. The differences between measured and calculated temperatures are very small for stable starting conditions of the natural convection. Unstable conditions with a flow reversal or very small mass flow rates (e.g. less than 0.1 kg/s) can cause numerical and/or physical instabilities.

3.2 Comparison with LOCA results

A loss-of-coolant accident is one of the most severe accidents for a nuclear power plant. To demonstrate the safety behavior incorporated into small HTR designs, LOCA simulation tests (no. HTA-5) have been conducted with the AVR reactor at KFA in Jülich. For the computer simulation of the LOCA test, dynamic heat transport analyses have been performed to determine the temperature distribution throughout the AVR reactor during the experiment. Different heat transport codes - like THERMIX-C-2D and HEATING-3D - have been used for pretest and posttest calculations in an international cooperation between KFA, JAERI, and ORNL.

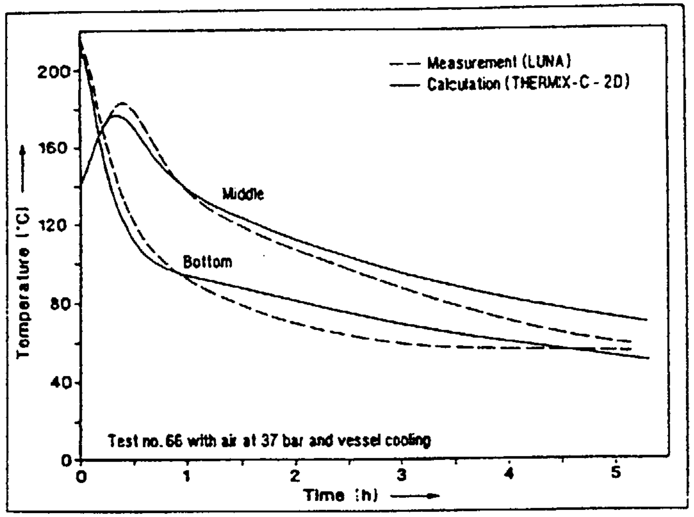

Fig. 11 · Pretest Temperature Calculation of the Pebble-Bed, Test with Natural Convection in the Core

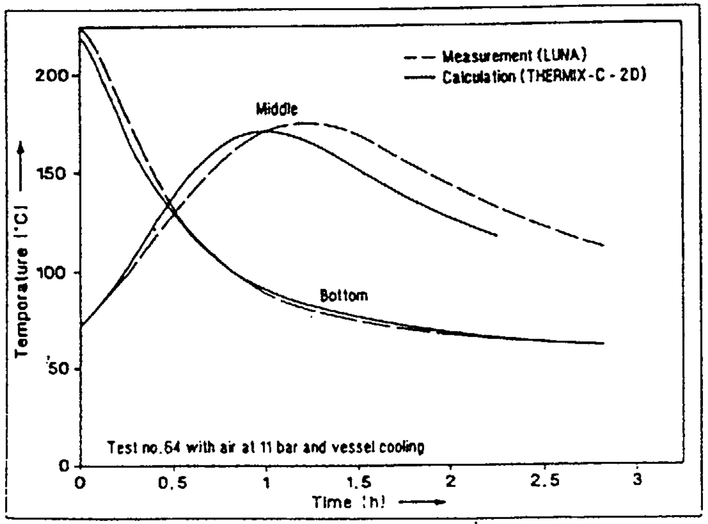

Fig. 12 · Posttest Temperature Calculation of the Pebble-Bed, Test with Natural Convection in the Primary Circuit

The measured core maximum temperature of 1080 °C was well predicted in pretest calculations, for example, performed by Interatom/Siemens (Fig. 13). Certain deviations occured for the position and the time of the core maximum temperature due to preliminary data for the LOCA test.
Our posttest calculation of the core maximum temperature (Fig. 14) is in very good agreement with the temperature recorded by the monitor elements. The accuracy of the 3-D HEATING analysis result is comparable with the interpretation of the accuracy of the experimental result within the range of +/- 8°C; the THERMIX-C-2D results show an accuracy of + 40 °C and - 20 °C depending on the modelling approach of the AVR core. This also means that the used correlation of the effective thermal conductivity for the pebble-bed core is correctly modelled.

For the licensing procedure predictions of the temperature in the core components were made by AVR/ORNL on a conservative basis /14/. The comparison showed significant differences in the component temperature behavior (Fig. 15) and in regions with 3-D temperature effects, especially in the reflector nose above the pebble-bed core.

In joint best estimate posttest calculations an expanded HEATING-3D model was used with updated input data. Improved 3-D calculations also took into account: natural convection heat transfer coefficients as boundary conditions due to natural convection circulations in horizontal gas spaces and vertical gas gaps in outer reactor structures

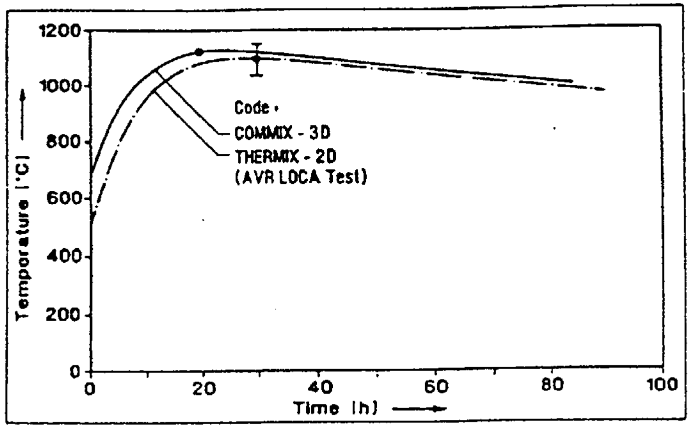

Fig. 13 · Pretest Core Max. Temperature Calculation (Ref. Interatom)

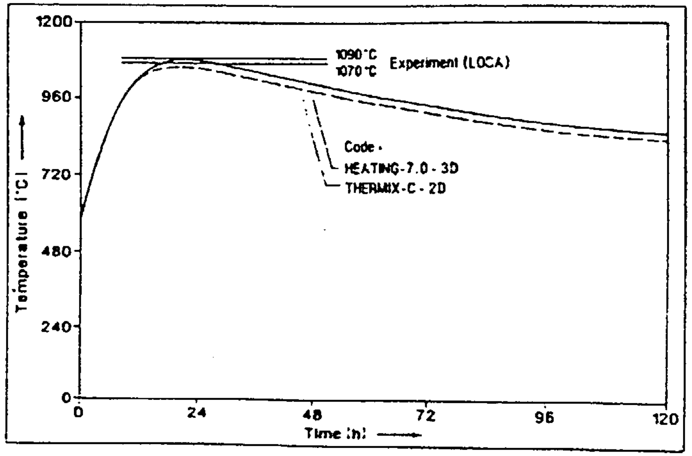

Fig. 14: Posttest Core Max. Temperature Calculation (Ref. KFA - ISR)

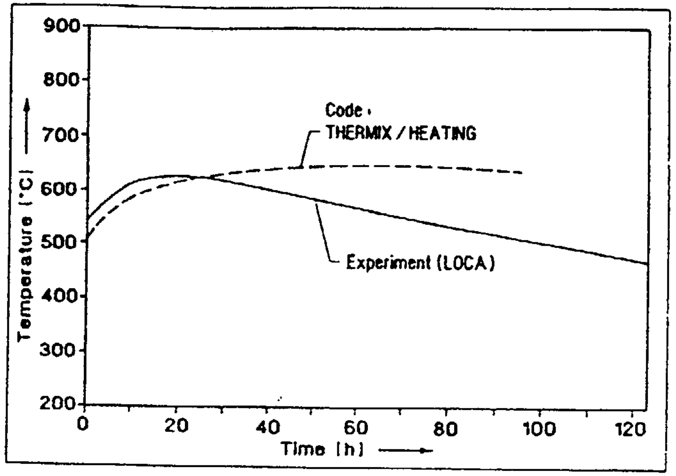

Fig. 15: Pretest Temperature Calculation of the Side Reflector (middle) (Ref. AVR / ORNL)

/15/. However, natural convection effects in the pebble bed core itself are not significant under LOCA conditions as the AVR tests with open and closed primary circuit shut-off-device have shown. This analytical approach showed a much better agreement with the experimental results (Fig. 16). Calculated temperatures are within the uncertainty of the measured temperatures of the components. However, azimuthal temperature effects and the heat flows to the heat sinks have to be analysed with a more sophisticated model which is underway.

The modelling of the LOCA test with the HEATING-3D code is meanwhile being tested for the LUNA-3D test facility with a detailed measurement. Azimuthal temperature effects can be produced in an annular pebble-bed by an electrical heating of the vessel wall in one of the six segments. First test results show the separate influence of natural convection, thermal radiation, and heat conduction in the segments (e.g. significant for local heat bridges in the primary system as the decay heat removal has shown for the THTR).

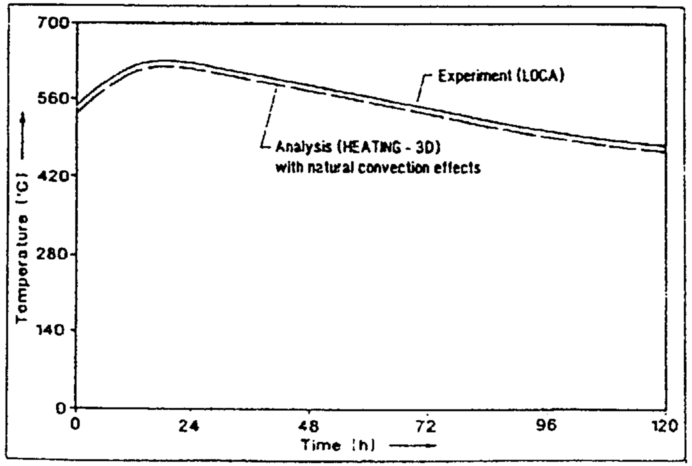

Fig. 16: Posttest Temperature Calculation of the Side Reflector (middle) (Ref. ISR / ORNL / JAERI)

4. CONCLUSION AND IMPORTANCE FOR THE SAFETY CONCEPT

Thermodynamic safety analyses have been performed for a modular HTR unit to investigate temperature loadings of safety barriers during severest accidents with passive heat removal dissipation into the reactor structures and ultimate heat sinks. In this context, the temperature response behavior has been analysed for postulated accident sequences associated with: anticipated transients without scram, loss-of-forced convection, loss-of-coolant, loss-of-vessel cooling, and primary system ruptures, regarding to sensitivities and safety margins. The following statements seem to be important for the safety concept:

- Thermodynamic methods have been proved to be suitable for accident analyses under loss-of-forced convection and loss-of-coolant simulation conditions. The comparison of analytical and experimental results implicates comparatively high accuracies for calculating self-limiting heat dissipation processes.

- Sensitivity analyses suggest greater safety margins concerning failure limits of safety barriers because of conservatisms, especially for the first barrier of the fuel element under LOCA condition. Reduced conservatisms of the safety margins might be incorporated into the design of a HTR module unit in the interest of optimizing the safety concept.

- This approach could be achieved by maximizing the power of the modular units together with an assembling in one closed containment to reduce economic costs (e.g. 1000 MWt power plant with 4 units of 250 MWt instead of 5 units of 200 MWt). Such a containment design would include additional safety aspects for unlikely events (e.g. large air/water ingress). A carbon brick layer above the concrete foundation could act as passive precaution for the core integrity.

For more rigorous safety criteria of future nuclear power plants - which are in discussion - with energy generation and process heat application in industrial areas, improved safety enclosures should assure an additional potential of the safety barriers in severest accident scenarios with very unlikely events almost by a deterministic approach. However, a comparative safety philosophy for improved containments of the next reactor generation with LWR and HTR should be considered in international cooperation based on PSA work to fulfill the question: How safe is safe enough?

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